NON-PUBLIC?: N
ACCESSION #: 9110180017
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Catawba Nuclear Station, Unit 1 PAGE: 1 OF 7

DOCKET NUMBER: 05000413

TITLE: Reactor Trip On Turbine Trip Due To Los Of Both Main Feedwater
Pumps On High Discharge Pressure
EVENT DATE: 09/11/91 LER #: 91-019-00 REPORT DATE: 10/08/91

OTHER FACILITIES INVOLVED: DOCKET NO: 05000

OPERATING MODE: 1 POWER LEVEL: 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR
SECTION:
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
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COMPONENT FAILURE DESCRIPTION:
CAUSE: SYSTEM: COMPONENT: MANUFACTURER:
REPORTABLE NPRDS:

SUPPLEMENTAL REPORT EXPECTED: No

ABSTRACT:

On September 11, 1991, at 13:02:52 hours, with Unit 1 in Mode 1, Power Operation, and operating at 100% Reactor power level, a Reactor (Rx) trip occurred as a result of a Turbine trip above P-9 (69% Rx power). The Turbine tripped because of loss of both Main Feedwater (CF) pumps on high discharge pressure. The high discharge pressure occurred when the Advanced Digital Feedwater Control System (ADFCS) sensed that there was a loss of two out of three Nuclear Power Channel inputs. The ADFCS responded properly by reducing the S/G level setpoint from 66.5% to 38% based on its median selected value of 0% Reactor power. The setpoint change caused the CF control and bypass valves to rapidly close. The valve movement caused the feedwater header pressure to increase, tripping the CF pumps on high discharge pressure and subsequently tripping the Turbine. This incident is attributed to a possible inappropriate action during nuclear instrument calibration, with a contributing cause of inappropriate action of failure to follow policies, directives, or

management procedures. Corrective action included addition of a four second delay in the activation of the ladder logic and input circuitry deviation alarms and 120 second lag time constant reducing the S/G program level to the ADFCS. The Independent Verification program is being evaluated to determine if training and program enhancements are needed.

END OF ABSTRACT

BACKGROUND

The purpose of the Excore Nuclear Instrumentation EIIS:JG! (ENB) System is to monitor Reactor EIIS:VSL! Core leakage neutron flux and generate appropriate trips and alarms for various phases of Reactor operations. The three separate overlapping ranges of Source Range, Intermediate Range, and Power Range also provide control functions and indicate Reactor status during Mode 2, Startup and Mode 1, Power Operation. Technical Specification 4.3.1.1 requires that channel calibration be performed daily on the Power Range Neutron Flux High Setpoint. This is to be performed by comparison of calorimetric (reactor thermal power best estimate, based on actual plant temperatures) to excore power (based upon nuclear power levels from excore instrumentation) when the Unit is above 15% Rated Thermal Power (RTP). Excore channel gains are to be adjusted to make indicated excore power consistent with indicated calorimetric power whenever this comparison reveals an absolute difference of more than 2% between the two.

Technical Specification 3.3.1, Table 3.3-1, requires that three out of four channels of Power Range Nuclear Instrumentation (PRNI) must be operable during Modes 1 and 2.

During Mode 1, a power range channel must be considered INOPERABLE whenever a mismatch exists between calorimetric power and excore power indication that is greater than 5.0% in the non-conservative direction (calorimetric power greater than excore power). If the mismatch is between 2.0% and 5.0% in the non-conservative directions, the channel is OPERABLE as long as the calibration process has been initiated. When the Unit is engaged in a power maneuver which results in a mismatch between calorimetric and excore power in excess of 2%, excore adjustment may be delayed until the Unit reaches a steady-state-power level, provided the mismatch does not exceed 5.0% in the non-conservative direction, as Specified by the Technical Specification Interpretation, dated November 1, 1990.

The Feedwater EIIS:SJ! (CF) system contains two feedwater pumps EIIS:P!. Normally, both pumps will be operating with each pump handling half of the feedwater flow. Feedwater pump high discharge pressure alarms in the control room, and high-high discharge pressure trips the associated pump with a twenty second time delay and a two out of three trip logic. Downstream of the feedwater pumps, the feedwater passes through two stages of high pressure feedwater heaters EIIS:HX! to a final feedwater header where the feedwater temperature is equalized. The feedwater is then admitted to the Steam Generators (S/Gs) through the four S/G feedwater lines EIIS:PSP), each of which contains a feedwater control valve EIIS:V) and a feedwater flow nozzle.

Feedwater flow to the individual steam generators is controlled by a three element feedwater control system EIIS:JB! using feedwater flow, steam generator water level, and main steam flow as control parameters for the steam generator feedwater control valves (1CF 28, 37, 46, and 55) and feedwater

control bypass valves (1CF30, 39, 48, and 57). Main feedwater pump speed is varied to maintain a programmed differential pressure (D/P) between the main steam EIIS:SB! (SM) header and the feedwater discharge header. The speed controller EIIS:XC! continuously compares the actual D/P with a programmed reference pressure (P-ref) which is a linear function of steam flow.

The Auxiliary Feedwater EIIS:BA! (CA) System assures sufficient feedwater supply to the steam generators in the event of loss of the Main feedwater. The system is designed to start automatically in the event of loss of offsite electrical power, trip of both CF pumps, safety injunction signal (SS), or low-low steam generator water level, any of which may result in, coincide with, or be caused by a Reactor trip.

The CA System consists of three auxiliary feedwater pumps, each powered from separate and diverse power sources. Two full capacity motor EIIS:MO! driven pumps are capable of supplying feedwater to two steam generators. These pumps will start automatically and provide flow within one minute following initiation of the system. Initiation conditions are any one or combination of the following: 2 of 4 low-low level alarms in any 1 of 4 S/Gs, loss of all CF pumps, initiation of a SS, or loss of offsite power.

In addition, a turbine EIIS:TRB) driven CA pump is capable of supplying feedwater to two S/Gs. The turbine driven pump will start automatically and provide flow on one or both of the following conditions: 2 of 4 low-low level alarms in any 2 of 4 S/Gs; loss of offsite power.

The Advanced Digital Feedwater Control System (ADFCS) provides functional design features. These features include an improved Steam Generator water level control system, signal validation to improve fault tolerance to sensor failures, and controller reliability. The steam generator water level control system is designed to provide automatic control without the need for operator intervention over the range of power operation. The feedwater system employs signal validation for input signals in order to reduce the probability of a failed sensor causing an upset condition in the plant. The system employs redundant controller outputs for all analog modulating control signals.

EVENT DESCRIPTION

On September 11, 1991, at approximately 1010 hours with Unit 1 operating in Mode 1, Power Operation, at 100% Reactor power level, W/R 56208OPS was issued to investigate the cause of a power range lower detector high flux deviation. Power Range Nuclear Instrument (PRNI) N-42 was placed in the bypass selection mode to allow maintenance to begin work.

At approximately 1300 hours, the"Instrument and Electrical (IAE) technician was returning N-42 PRNI to service using procedure IP/1/A/3240/04I, Power Range - N42 Analog Channel Operational Test. At approximately the same time, a downward trend occurred on the N-41 PRNI output to the Operator Aid Computer

OAC) and the ADFCS. The downward trend on N-41 PRNI combined with the N-42 PRNI, indicating 0% Reactor power, caused the ADFCS to respond as designed to 2 out of 3 channels from the Nuclear Instrumentation System (NIS) panel deviating greater than 10% Reactor power.

At 13:02:49 hours, the ADFCS immediately started to close the CF control and bypass valves rapidly to respond to the lower S/G setpoint level based on the median selected value of less than 100% Reactor power. The valve movement, with both feedwater pumps at 100% output, caused the CF header pressure to increase rapidly and exceed the feedpump high discharge setpoint, which caused both CF pumps to trip. The loss of both CF pumps caused this Turbine to trip. At 13:02:52 hours, the Reactor tripped as a result-of the Turbine trip above P-9 (Reactor power greater than 69%).

Following the Reactor trip, the Unit was stabilized in Mode 3, Hot Standby. Both Motor Driven Auxiliary Feedwater (CA) pumps auto started on loss of both CF pumps. During the CF pressure transient, the CF pump discharge pressure exceeded 1441 psig. This pressure transient caused the shell side relief valves on feedwater heaters 1A1 and 1A2 to open. During the post-trip review, Performance noted that 1SB-003, Condenser Steam Dump Valve, did not close after the transient in a timely manner. Work requests were written to determine if the feedwater heater relief valves were damaged and why 1SB-003 did not close in a timely manner as required.

CONCLUSION

This incident is attributed to an unknown cause with a possible inappropriate action during NI calibration, with a contributing cause of failure to follow. policies, directives, or management procedures. At the time of the Reactor trip, NIS channel deviation calibration of N-42 PRNI was in progress. After completing the maintenance activities, the IAE technician was preparing to return N-42 back to service. When completing step 10.4.10 of IP/1/A/3240/04I, the IAE technician indicated that he inadvertently operated the N-41 rod stop bypass switch without allowing the independent verifier to perform the double verification.

The root cause of possible inappropriate action was assigned because manipulation of the N-41 rod stop bypass switch, the one the IAE technician said he manipulated, would not have caused this incident. Operation of the N-41 power range mismatch switch would have caused the incident. The contributing cause of failure to adhere to policies, directives, or management procedures was assigned because the IAE technicians did not perform the IV as directed by these policies. Proper use of IV may have prevented this incident.

Maintenance Engineering and Services (MES) investigated the implication of mispositioning the rod stop bypass switch by recreating the event, and determined it would have no effect on the NIS inputs to the ADFCS. It should

be noted that the downward trend on N-41 occurred at about the same time that the IAE technician indicated that he switched the wrong N-41 rod stop bypass switch. With the N-42 power range mismatch bypass switch still in bypass and the N-41 power mismatch selector switch operated to bypass, the N-41 downward trend could have been explained.

As a result of the increased pressure on the feedwater heaters, 1A1, 1A2, 1B1, and 1B2, shell side, relief valves 1CF-64, 86, 71, and 74 opened, and two welds experienced some damage. The valves were replaced under

W/R 91074320, 91074317, 91074314, and 91074323. The two welds were cut out and rewelded per W/R 52232OPS and 56223OPS-1.

During the Abnormal Plant Event (APE) meeting, the Performance Engineer noted that valve 1SB--003 did not close in a timely manner. W/R 8039PRF was written to investigate. Also, questions were raised as to what additional administrative controls are needed for other "system" related software. At present, changes to software like the ADFCS software packages are outside the scope of station directive requirements. Projects will evaluate and recommend any needed changes to existing directives.

A search of the Operator Experience Program (OEP) data base for the past 24 months revealed two Reactor and main turbine-trips as a result of a main feedwater pump trip. LERs 414/90-013 and 413/91-015 describe events attributed to Design/Construction/Installation Deficiencies and Equipment Failure. Because of the difference in root causes, this event is not considered to be recurring under Nuclear Safety Assurance Guidelines.

CORRECTIVE ACTION

IMMEDIATE

1) Control Room operators entered EP/1/A/5000/01, Reactor Trip or Safety Injection, and EP/1/A/5000/01A, Reactor Trip Response, and recovered Unit 1 to normal Mode 3, Hot Standby, condition.

2) IAE technicians returned N-41 Rod Stop Bypass Selector Switch to operate.

SUBSEQUENT

1) Control Room operators entered OP/1/A/6100/05, Unit Fast Recovery, in preparation for restarting the Unit.

2) MES changed the software in the ADFCS to make the system more tolerant to an input spike or random noise. A timer was added to each NIS signal, set at 4 seconds, to delay the activation of the ADFCS ladder logic and input circuitry deviation alarms.

3) MES changed the S/G level program setpoint from a 5 second lag "time constant" to 120 seconds. This will reduce the speed of the S/G level program ramping to its new setpoint.

4) CF relief valves and some associated piping were replaced under work requests 55232OPS, 56223OPS-1, 91074320, 91074317, 91074314, and 91074323.

5) Work request 8039PRF written to investigate the cause of the slow response of 1SB-003 when closing.

PLANNED

1) IAE independent verification procedures and guidelines will be reviewed to determine if they are adequate.

2) Operations and IAE will enhance NIS procedures by including in the procedures to "Force" the NIS output to its "as-found" value within the ADFCS prior to placing the NIS channel to bypass. This change will allow the testing of the NIS system without degrading the median selected input to the S/G level program portion of the ADFCS system. Listed below are those procedures that will be enhanced:

a) IP/1/A/3240/11 Calibration Procedure NIS Power Range Calibration at Power

b) IP/1/A/3240/12 Removing NIS Channels from Service

c) IP/0/A/3240/14 Excore Nuclear Instrumentation System (Incore Excore Calibration)

d) IP/1/A/3240/17 Overpower Trip High Range Setpoints

e) IP/1,2/A/3240/04H,I,J Power Range N-41, N-42, and N-43 Analog Channel Operational Test

f) IP/1/A/3240/04C Excore Nuclear Instrumentation System (ENB) 'Power Range Channel Calibration

g) OP/1/A/6100/21 Operations Calibration Procedure for Power Range NIS

3) Projects will evaluate the ADFCS software and determine if changes to the software systems are controlled under the current Station Directives.

SAFETY ANALYSIS

At the time of this incident, Unit 1 was operating at 100% power. The Reactor trip occurred due to CF feed pump high discharge pressure trip and a subsequent Turbine trip. Following the Reactor trip, Reactor power immediately decreased to zero. No primary or secondary power operated relief valves or code safety valves were lifted during the event.

Reactivity was controlled by the Reactor trip. All rods EIIS:ROD! inserted, as expected. Residual heat was removed from the Reactor to the ultimate heat sink via CA.

A Reactor trip on loss of both main feedwater pumps is bounded by the "Loss of Normal Feedwater" transient described in Section 15.2.7 of the Catawba FSAR.

The health and safety of the public were not affected by this event.

ATTACHMENT 1 TO 9110180017 PAGE 1 OF 1

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DUKEPOWER

October 8, 1991

Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Subject: Catawba Nuclear Station
Docket No. 50-413
LER 413/91-19

Gentlemen:

Attached is Licensee Event Report 413/91-19, concerning REACTOR TRIP ON TURBINE TRIP DUE TO LOSS OF BOTH MAIN FEEDWATER PUMPS ON HIGH DISCHARGE PRESSURE.

This event was considered to be of no significance with respect to the health and safety of the public.

Very truly yours,

J. W. Hampton
Station Manager

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